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 FACIL: 50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269
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 RECIPIENT NAME RECIPIENT AFFILIATION

SUBJECT: LER 95-001-00: on 950106, potential unanalyzed main steam line break scenario occurred. Caused by single pipe break downstream of main steam branch isolation valves. Procedures for normal operation revised. W/950208 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 10
 TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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DUKE POWER

February 8, 1995

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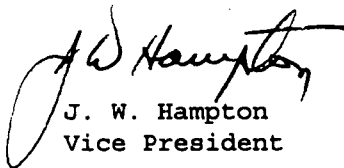
Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Voluntary LER 269/95-01

Gentlemen:

This Voluntary Licensee Event Report (LER) 269/95-01, concerning a potential unanalyzed main steam line break scenario, is provided for your information.

This condition is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,


J. W. Hampton
Vice President

/ftr

Attachment

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LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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| FACILITY NAME (1) Oconee Nuclear Station, Unit 1 | | DOCKET NUMBER (2) 05000 269 | PAGE (3) 1 OF 9 |
|--|--|---------------------------------------|---------------------------|

TITLE (4)
Potential Unanalyzed Main Steam Line Break Scenario

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT NUMBER (7) | | | OTHER FACILITIES INVOLVED (8) | |
|----------------|-----|------|----------------|-------------------|-----------------|-------------------|-----|------|-------------------------------|---------------|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| 01 | 09 | 95 | 95 | -- 01 -- | 00 | 02 | 08 | 95 | Oconee, Unit 2 | 05000 270 |
| | | | | | | | | | Oconee, Unit 3 | 05000 287 |

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| OPERATING MODE (9) N | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11) | | | |
| POWER LEVEL (10) 100 | <input type="checkbox"/> 20.402(b) | <input type="checkbox"/> 20.405(c) | <input type="checkbox"/> 50.73(a)(2)(iv) | <input type="checkbox"/> 73.71(b) |
| | <input type="checkbox"/> 20.405(a)(1)(i) | <input type="checkbox"/> 50.36(c)(1) | <input type="checkbox"/> 50.73(a)(2)(v) | <input type="checkbox"/> 73.71(c) |
| | <input type="checkbox"/> 20.405(a)(1)(ii) | <input type="checkbox"/> 50.36(c)(2) | <input type="checkbox"/> 50.73(a)(2)(vii) | <input checked="" type="checkbox"/> OTHER |
| | <input type="checkbox"/> 20.405(a)(1)(iii) | <input type="checkbox"/> 50.73(a)(2)(i) | <input type="checkbox"/> 50.73(a)(2)(viii)(A) | (Specify in Abstract below and in Text, NRC Form 366A) |
| | <input type="checkbox"/> 20.405(a)(1)(iv) | <input type="checkbox"/> 50.73(a)(2)(ii) | <input type="checkbox"/> 50.73(a)(2)(viii)(B) | |
| | <input type="checkbox"/> 20.405(a)(1)(v) | <input type="checkbox"/> 50.73(a)(2)(iii) | <input type="checkbox"/> 50.73(a)(2)(x) | |

LICENSEE CONTACT FOR THIS LER (12)

| | |
|---|---|
| NAME L.V. Wilkie, Safety Review Manager | TELEPHONE NUMBER (include Area Code) (803) 885-3518 |
|---|---|

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
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SUPPLEMENTAL REPORT EXPECTED (14)

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| YES (If yes, complete EXPECTED SUBMISSION DATE) | <input checked="" type="checkbox"/> NO | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

ABSTRACT

This is a voluntary Licensee Event Report. On January 6, 1995, the NRC notified Oconee Nuclear Station that it felt an Unreviewed Safety Question existed. The issue was the potential blow down of both steam generators due to a single pipe break downstream of main steam branch isolation valves which have been shown to be unable to close against the expected differential pressure. This was contrary to statements in the Oconee FSAR. On January 9, 1995, a conference call was held with the Office of Nuclear Reactor Regulation and Region II to discuss the NRC concern. After the call, at 1450 hours, with all three Oconee units at 100 % full power, Oconee made a one hour notification as a conservative measure. Operations procedures for normal operation were revised to close one isolation valve in each affected branch line that isolated one main steam line from the potential break. Duke will perform a Probabalistic Risk Analysis to evaluate if this is the safest operating mode, and will revise procedures and/or the FSAR, as appropriate, based on the results.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

BACKGROUND

The Main Steam (MS) System [EIIS:SB] for each Oconee unit consists of two main steam lines, each connected to one steam generator [EIIS:SG], associated relief valves [EIIS:RV], turbine stop valves, and several branch lines which provide flow paths to the turbine by-pass valves, the Auxiliary Steam (AS) [EIIS:SA] header, the turbine driven emergency feedwater [EIIS:BA] pump turbine (EFWPT), etc. These MS branch lines are cross-connected such that one or both of the main steam lines can supply the line. See the MS Flow diagram attached. Oconee does not have Main Steam isolation valves.

Valves MS-24 and MS-33 (MS line A and B to Aux Steam) are motor operated valves which provide capability to isolate the two main steam lines from the cross connection header which supplies the AS Header, which can also be supplied from any other operating Oconee unit. MS-82 and MS-84 (MS line A and B to EFWPT) isolate the MS lines from the cross connection header which supplies the EFWPT steam supply header. In each case, the piping joins to a crossover header downstream of the valves. These valves are supplied from normal (non-Class 1E) power.

EVENT DESCRIPTION

In October 1992, testing resulting from NRC Generic Letter 89-10 determined that valves MS-24, MS-33, MS-82, and MS-84 would not close against differential pressures greater than 400 Psid. An entry into the Problem Investigation Process (PIP) was initiated to evaluate and track this problem.

On November 11, 1992, an operability evaluation was completed which determined that the valves were still considered operable and, therefore, no 10CFR 50.59 evaluation was considered necessary.

In Inspection Report (IR) 93-30, dated December 22, 1993, the NRC Site Residents made the inability of these valves to close against higher differential pressures an "Unresolved Item." The issue was that failure of these valves to close following a downstream line break could result in blow down of both steam generators (SG).

Based on an analysis received from Duke, the Site Residents included a "Notice of Deviation" in IR 93-31, which stated that Oconee was deviating from statements in the current Final Safety Analysis Report (FSAR) Sections 10.3.2 and 10.3.4 which indicate that the arrangement of valves and parallel piping prevents blow down of both SGs from a single leak in the system.

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On February 24, 1994, Duke responded to the Notice of Deviation, agreeing that there was a deviation from the FSAR and that Section 10.3.2 would be revised. Duke stated that the consequences of a break in the steam header piping downstream of these valves is bounded by the existing FSAR analyses.

In March 1994, NRC Region II issued a Task Interface Agreement to the Office of Nuclear Reactor Regulation (NRR) asking whether the postulated event involves an Unreviewed Safety Question (USQ) in that it represents the possibility for an accident or malfunction of a different type than any evaluated in the FSAR. On April 8, 1994, Duke provided to NRR an analysis of the auxiliary steam header break. This included a statement that no USQ exists. The analysis did not include any references to prior documents related to this issue.

On Friday, January 6, 1995, the NRC issued a letter stating that they considered the auxiliary steam line break issue to be an USQ, and stated that it should be reported under 10CFR 50.72(b)(1)(ii)(B) as being outside the design basis of the plant.

On Monday, January 9, 1995, Oconee management discussed the January 6 NRC letter during a conference call with NRR and Region II. During this conference call, Oconee management indicated that additional review of the NRC letter would be needed to determine if Duke concurred that an USQ existed. However, Oconee agreed to conservatively make a 50.72 notification and to take compensatory action by closing one of the affected valves on each common header on all three units.

The 50.72 notification occurred at 1450 hours. At the time of the notification, all three Oconee units were operating at 100 % full power. Appropriate operational procedures were revised to close one isolation valve in each affected branch line that isolated one main steam line from the potential break.

During the Oconee review of this issue, additional relevant documentation was identified. On December 19, 1972, the Atomic Energy Commission (AEC) issued Supplement 2 to the Safety Evaluation Report for Oconee Unit 1. Appendix E of that supplement addressed requirements for information needed to consider the effects of piping system breaks outside containment. In response, Duke provided Report OS-73.2, "Analysis of Effects Resulting From Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2 & 3", dated April 25, 1973. This report has also been referred to as the "High Energy Line Break Analysis Report."

In that report, Table 2.3-1 identified the postulated break locations. Two cases (1.c and 1.d, Main Steam breaks in the turbine building) indicated "possible loss of 2nd Main Steam line branch lines." Section 3.1.1

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addressed the Main Steam System line break scenarios. An analysis starting on page 3.1-2 of that report stated that these two cases "identify main steam line breaks which could possibly damage branch lines terminating in the adjacent main steam line. Consequently, this could lead to a double steam generator blow down." The analysis of this accident concluded that the reactor can be safely shut down and "the operator would not need to take immediate action other than verify that the automatic actions have occurred. The operator within a few minutes will take action to regulate feedwater to steam generators to affect an orderly cooldown." This statement specifically addresses the issue of restoring heat transfer to two steam generators which have blown down.

Subsequently, the AEC issued Supplement 1 to the Safety Evaluation Report for Oconee Nuclear Station Units 2 and 3, on July 6, 1973. Section 7.1.11, "High Energy Line Rupture External to the Reactor Building", Page 7-43, addressed OS-73.2 as follows:

"Report No. OS-73.2 summarizes the applicant's review of the consequences of postulated high energy line breaks with regard to environmental effects and physical damage to the station on a case-by-case basis. Contained in this summary are the operational analyses which describe the sequence of events following a piping break including the resultant reactor and primary transients. The cases considered are listed in OS-73.2 with engineering data, consequences, environmental effects, station situation, remedial action and required station modifications (if any)."

Page 7-44 contains the Staff Evaluation and Conclusion as follows:

"The staff has evaluated the assessment performed by the applicant and has concluded that the applicant has analyzed the facilities in a manner consistent with the intent of the criteria and guidelines provided by the staff. The staff agrees with the applicant's selection of pipe failure locations and concludes that all required accident situations have been addressed appropriately by the applicant."

Duke Power has also identified two other examples of analyses addressing the blow down of two steam generators.

- A. In response to IN 79-22 and a September 17, 1979 letter from the NRC, Duke performed an evaluation of potential non-safety-grade control system interactions during design basis high energy line break accidents. The Duke evaluation was submitted to the NRC by a letter dated October 5, 1979. The evaluation addressed the inadvertent opening of a Turbine By-pass Valve (TBV) during a steam line break.

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The letter states that this scenario would result in additional blow down of the steam system and referenced the High Energy Line Break Analysis Report analysis of the blow down of both steam generators as bounding the open TBV scenario. Although the NRC did not directly respond to this submittal, NUREG-0649 stated that all licensees responded to IN 79-22 and that the NRC screened the responses. The NRC review concluded that no specific event leading to unacceptable consequences was identified.

B. In 1984, the Oconee Standby Shutdown Facility (SSF) [EIS:NB] was made operational. This facility was designed to maintain hot shutdown conditions on all three Oconee units for up to 72 hours in response to fire, flood, and/or security events. The facility and associated systems include an independent diesel generator for power, an Auxiliary Service Water pump to provide emergency feedwater to all three units, and a Reactor Coolant Make-up Pump for each unit to compensate for reactor coolant pump seal leakage and Reactor Coolant volume decrease due to temperature/density changes.

On July 9, 1986, Duke submitted to the NRC a B&W analysis that concluded that the SSF can mitigate the consequences of a double steam line break. This analysis uses slightly different assumptions than the current FSAR Section 15.13, the most significant difference being the components available to mitigate the scenario. It assumes that the High Pressure Injection and Emergency Feedwater systems are unavailable and uses the SSF Reactor Coolant Makeup Pump and SSF Auxiliary Feedwater Pump instead. This analysis also assumes a break area that is an order of magnitude larger than the break sizes mentioned in the NRC January 6, 1995 letter.

From the time that Oconee Unit 3 received its operating license until the late 1970s, the FSAR was not maintained current. In 1980, the Oconee FSAR was rewritten and reissued as a "living document" and has been updated annually to reflect changes in the plant and to address appropriate emerging issues.

CONCLUSIONS

This is a voluntary Licensee Event Report.

Based on the statements in OS-73.2 and Supplement 1 of the Safety Evaluation Report for Oconee Units 2 and 3, it appears that the possibility of a Main Steam (MS) line break resulting in blow down of both steam generators (SGs) has been reviewed and the consequences were found to be

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acceptable to the AEC under the criteria and guidelines in effect at the time of that review.

Unfortunately, previous submittals related to the Notice of Deviation did not provide a historical perspective on analyses which have been submitted to the NRC regarding the consequences of blowing down both SGs.

In response to the January 6, 1995 letter, an immediate corrective action was isolating the affected cross connection steam headers from one SG. This was accomplished by closing one of the affected valves in each cross connection header. Although this action precludes a hypothetical blow down of both SGs, it also reduces the number of steam supplies to the Turbine Driven Emergency Feedwater Pump. Therefore, Duke Power is concerned that this may not be the configuration with the lowest impact on overall safety.

As a result, Duke Power will perform a Probabalistic Risk Analysis of significant options to determine the safest configuration of the MS system for normal operation. After appropriate consideration, Duke will perform a 10CFR50.59 evaluation, implement the "safest" option, and revise the FSAR as necessary.

Oconee Nuclear Station has had a Design Basis Documentation (DBD) project in progress since 1989. It is currently expected to be completed in 1995. The issue of the deviation from the FSAR arose prior to the start of the Main Steam System DBD. However, it is expected that the deviation would have been detected and resolved as part of the MS DBD preparation and review. Several Licensee Event Reports have been submitted to document problems found on other systems during the course of this project. In that respect, this report is considered recurring, but no corrective action from previous events could have prevented this condition.

There were no NPRDS reportable equipment failures, personnel injuries, or releases of radioactive materials associated with this report.

CORRECTIVE ACTIONS

Immediate

1. Upon completion of the teleconference of January 9, 1995, Operations revised appropriate procedures on each Oconee unit to isolate each cross connection header from one steam generator by closing one of the affected valves in each header.

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Subsequent

1. Duke initiated a documentation review to determine past AEC/NRC positions on this issue.

Planned

1. Duke Power will perform a Probabalistic Risk Analysis evaluation of potential system configurations and determine the most appropriate (safest) long term solution.
2. Implement the long term solution, including FSAR revisions, as appropriate.

SAFETY ANALYSIS

The AEC safety evaluation report concluded that the results of a blow down of both steam generators (SGs) were acceptable.

There are two affected branch headers, the Turbine Driven Emergency Feedwater Pump Turbine (TDEFWPT) steam supply and the Auxiliary Steam (AS) supply header. A break on either of these lines, downstream of the isolation valves, would permit both SGs to blow down. The valves can be closed manually but manual closure is not assumed possible until after the blow down is completed. An additional branch line from each Main Steam (MS) line supplies steam to the associated main feedwater pump, but the blow down will affect that source also. A start-up steam source is supplied to the main feedwater pumps from the AS header, however that source could be affected if the break is in the AS supply header and the affected unit happens to be the one supplying the AS header. Therefore, there could be a total loss of steam for the TDEFWPT and main feedwater pumps on the affected unit. (If one of the other two Oconee units is supplying the AS header or is aligned by operator action after the break, the main and emergency feedwater pumps could continue to be available throughout the event.)

According to OS-73.2, a break at either location could result in the Auxiliary Feedwater Pump being used to restore feedwater some minutes after the SGs had boiled dry. Operations procedures specify that flow would be reestablished cautiously, using "trickle" feed (i.e. intermittent low flow rates), to reduce the impact of thermal stresses on the SGs.

After the submittal of OS-73.2, Oconee enhanced the ability to provide feedwater to the SGs. Oconee has added two Motor Driven Emergency Feedwater pumps (MDEFWP) per unit. Also, the capability exists to cross

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connect the Emergency Feedwater (EFDW) headers of all three Oconee units such that the affected unit could be fed from another Unit's EFDW pumps. Also, a completely separate Stand-by Shutdown Facility (SSF) includes an additional Auxiliary Feedwater Pump which is capable of supplying feedwater to all three Oconee units simultaneously, powered by a diesel generator independent of all other site AC power.

If, for some reason, it became impossible or undesirable to feed the SGs, Operations could use "feed and bleed" (also known as High Pressure Injection (HPI) [EIIS:BG] Forced Cooling) to control Reactor Coolant System (RCS) [EIIS:AC] temperatures by using HPI flow to establish flow through the Pressurizer Power Operated Relief Valve. This is a less desirable method, but procedural guidance is provided as a contingency.

Therefore, the ability to adequately provide core decay heat removal has been demonstrated for all scenarios associated with this condition and the health and safety of the public were not affected by this condition.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Attachment A

Summary Flow Diagram
Main Steam System

